Tokamak Experiments

NSTX	H. Kugel, C. Skinner
LTX	D. Majeski
C-MOD	D. Whyte
DIII-D	A. McLean
DiMES/MiMES	D. Rudakov

PSI Program e-conference meeting
November 2nd, 2005

Supported by Office of Science

Recycling Measurements Following Repeated Lithium Pellet Injection

H. W. Kugel, M. Bell, T. Gray, D. Mueller, B. LeBlanc, R. Kaita, T. Stevenson, C. H. Skinner, A. L. Roquemore (PPPL), C.Bush,

R. Maingi (ORNL), V.Soukhanovskii (LLNL), R. Raman (UWa)

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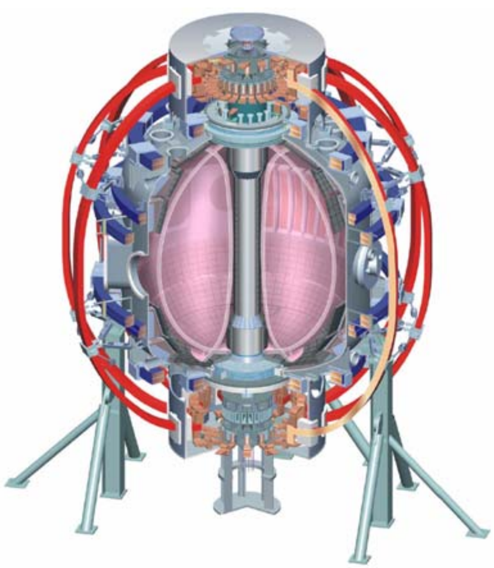

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NSTX Has Been Investigating Lithium Pellet Injection for Reduced Recycling as Part of a 3 Phased Approach to Lithium PFCs: (I) Li Pellet Injection, (II) Li Evaporators, (III) Liquid Li Divertor

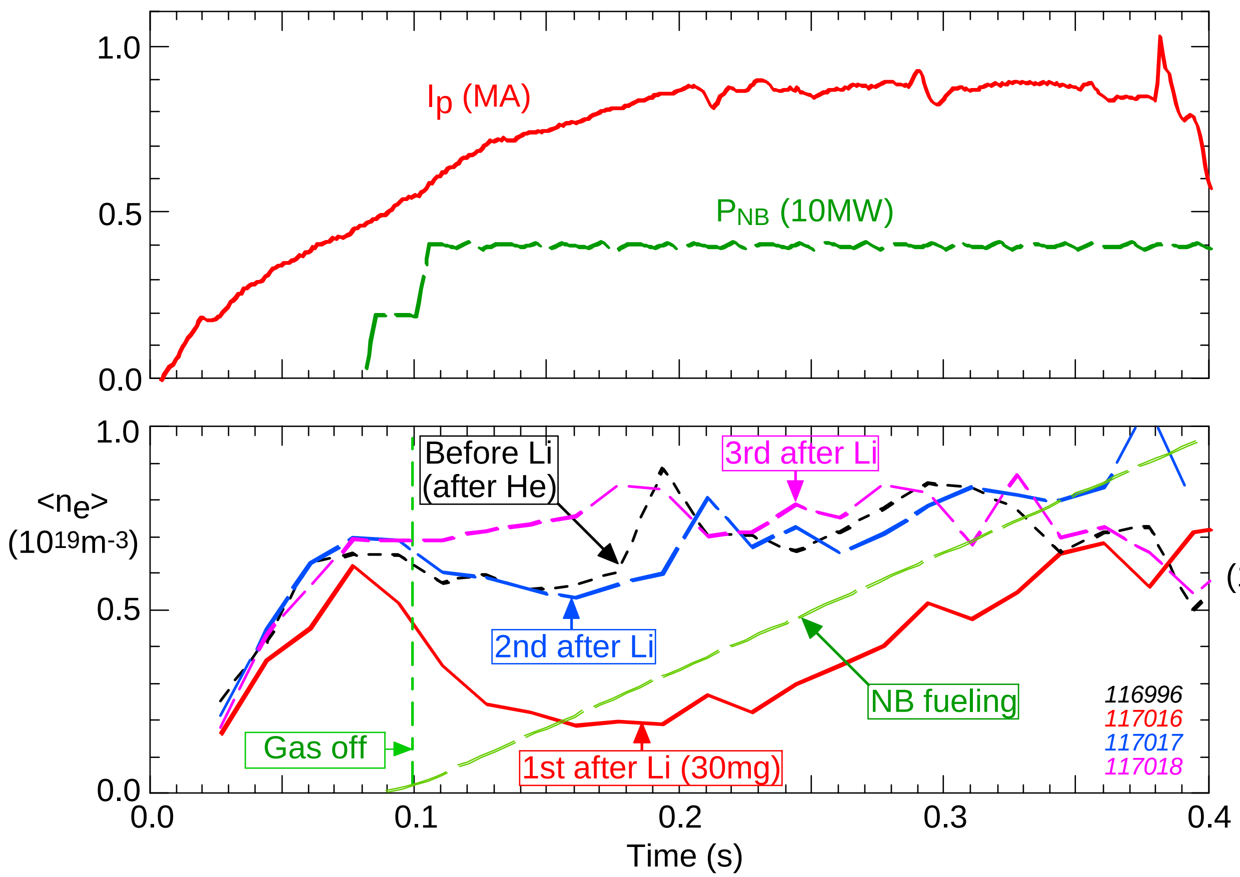
• TFTR obtained reduced recycling and significantly enhanced performance by starting with a thoroughly degassed limiter and applying lithium deposition techniques directly into low density plasmas.

• Since TFTR, Lithium Pellet Injection was applied directly into normally fueled, diverted C-MOD, DIII-D, TdeV, and NSTX plasmas, but without thorough wall degassing, and has yielded no similar significant performance improvement other than a small decrease in impurities.

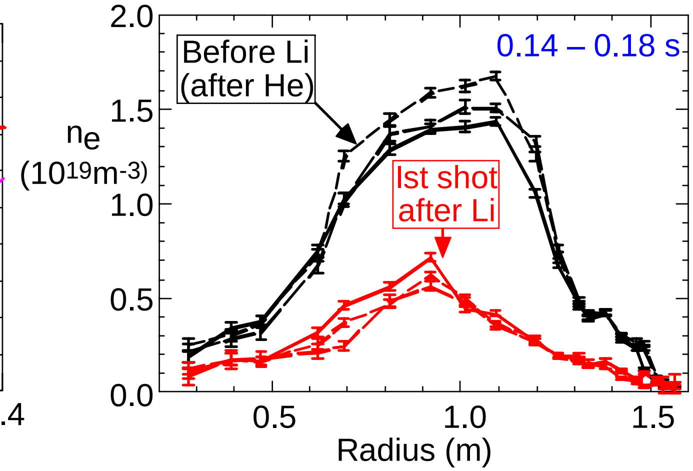
• The goal of these NSTX experiments was to make contact with the TFTR lithium experience, starting with the recycling effect.

• These experiments investigated recycling, first from the *NSTX Inner Toroidal Limiter* (Center Stack), and then from the *Lower Divertor* following repeated lithium pellet injection.

Exp-1: Initial CSL NBI Deuterium Reference Shot Following 30 mg of Lithium Deposition on CS Exhibited ~x3 Decrease in Density and Peaked Profiles

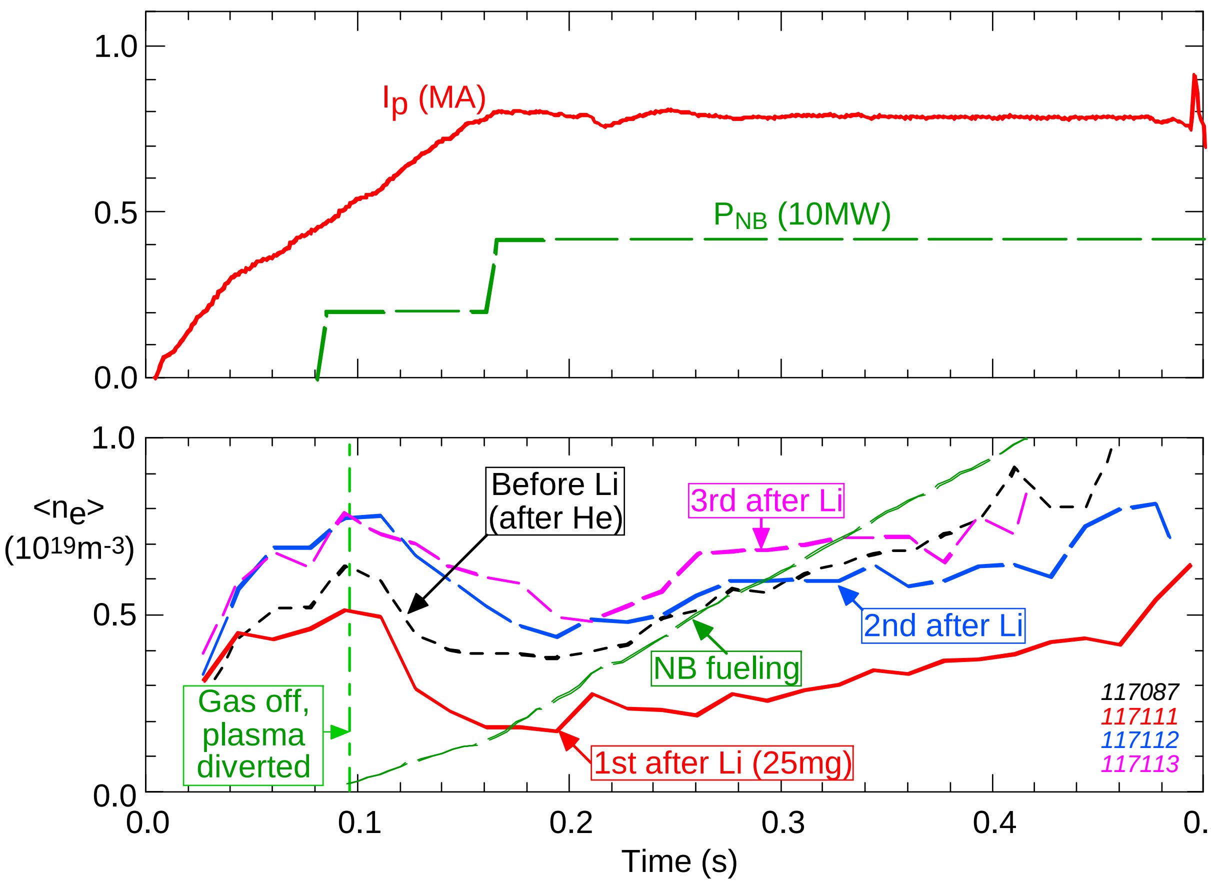


Center-stack limiter discharges,
0.9 MA, 0.45T,
D_2 gas fueling 3.5mg

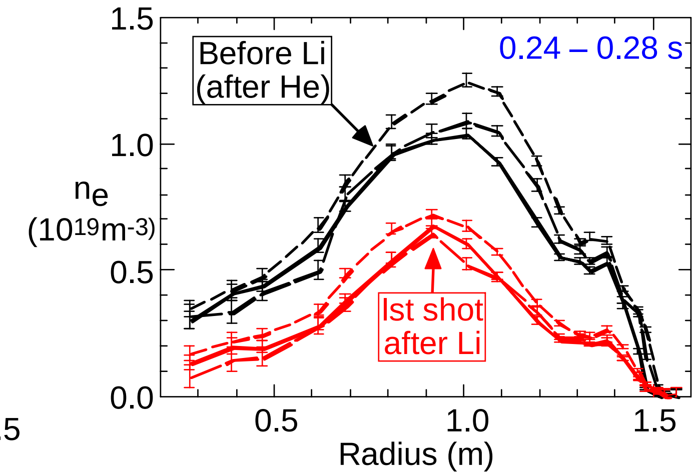


- 30 mg of lithium pumping of edge density saturated after the 3 Reference Discharges and returned to pre Li wall conditions.
- Exp-2 using an additional 24 mg Li duplicated these a CSL results.

Exp-3:LSN NBI D Reference Shot Following 25 mg of Li Deposition on Lower Divertor Exhibited ~x2 Decrease in Density and Peaked Profiles



Lower single-null divertor discharges, 0.45T, D_2 gas fueled 3.5mg



- 25 mg of lithium pumping of edge density saturated after the 3 Reference Discharges and returned to pre Li wall conditions.
- Expected if most injected gas reacts with the deposited lithium

H. Kugel

Conclusions

- The results are consistent with the consumption of the deposited lithium.

  E.g. ~30 mg Li = 2.6×10^{21} Li atoms available to react with 2.6×10^{21} D

  ~9×10^{20} D/Shot, and Li pumping stops ~2-3 shots ($1.8\text{-}2.7 \times 10^{21}$ D removed)

- The CSL recycling results made contact with the TFTR lithium recycling experience.
- The LSN results extended the TFTR lithium recycling experience to a diverted configuration.
- LPI directly into LSN plasmas yielded no pumping effect (similar to previous NSTX and other diverted results).
- NSTX Phase I (Li Pellet Injection) experiments demonstrated that surfaces ***pre-coated with lithium***, <u>edge pumped</u> a diverted plasma and exhibited an increased <u>peaking of the density profile</u>.
- NSTX Phase II (Lithium Evaporator) is in preparation for performing routine thick lithium coating depositions over a significant fraction of the plasma facing surfaces for the first Experimental Proposals in early 2006.

Supported by Office of Science

NSTX

Time resolved measurements of deposition and dust in NSTX

C.H. Skinner, H. Kugel, L. Roquemore, E. Biewer, W. Davis, R. Maingi, N. Nishino, C. Parker, and C. Voinier

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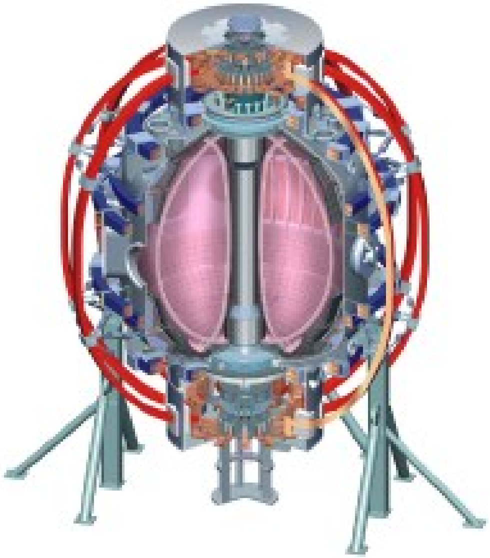

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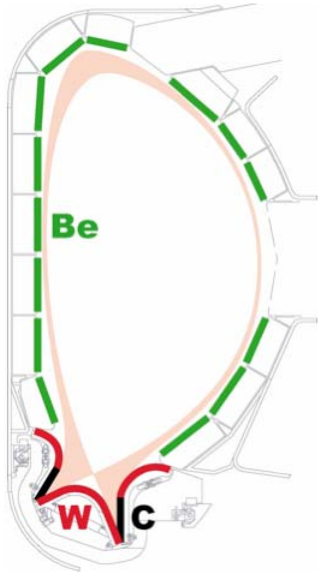
U Quebec

Motivation:

- **ITPA DSOL priority topic: "Improve understanding of SOL plasma interaction with the main chamber"**
  Concerns on ITER Be wall:
  - **heat load, erosion lifetime, repair difficulty**
  - **tritium migration,**
  - **coating diagnostic mirrors…**

- **ITPA Diagnostics priority topic: "…assessment of techniques for measurement of dust and erosion."**
  Concerns: high dust levels are expected in ITER from long plasma duration and more intense plasma surface interactions.
  - **How to assure dust levels are below safety limit ?**
  - **Will dust transport impurities to plasma core reducing fusion reactivity ?**

Open ST geometry facilitates diagnostic access and cost effective progress on these topics.



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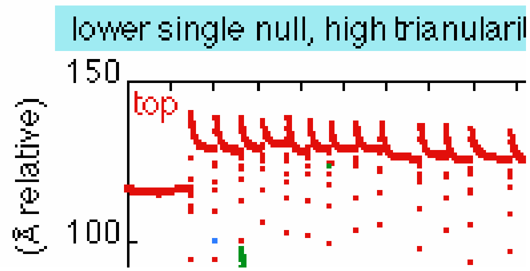
Dust	Safety Issue	Limits (kg)
Beryllium	Reactivity with steam Toxic	10-20 on hot surfaces
Carbon	Tritium retention Explosion with air	~100
Tungsten	Activation	100-400

•Limits for C-and Be-dust are related to an explosion (e.g., H produced by Be reactivity with steam).

•The limit for W-dust is related to the containment function of the ITER building (is more flexible).

Conclusions:

- Incandescent particles with complex trajectories observed with fast camera in some NSTX plasmas.
- Carbon dust identified on surfaces after campaign.
- Electrostatic dust detector developed for time resolved measurements, however more sensitivity needed for NSTX dust levels (should not be a problem in ITER).
- Quartz microbalance (qmb) show boronization is non uniform, movable glow probe installed to address this.
- Quartz microbalance shows erosion and deposition on wall from plasmas.
  - Deposition dominates on first discharge of day
  - Erosion depends on plasma shape.
  - Analysis continuing…



More in poster GP1.00025
"Controlling surface dust in a tokamak" Colin Parker.

LTX

Lithium pumping, improved confinement and reduced recycling

Dick Majeski

PPPL

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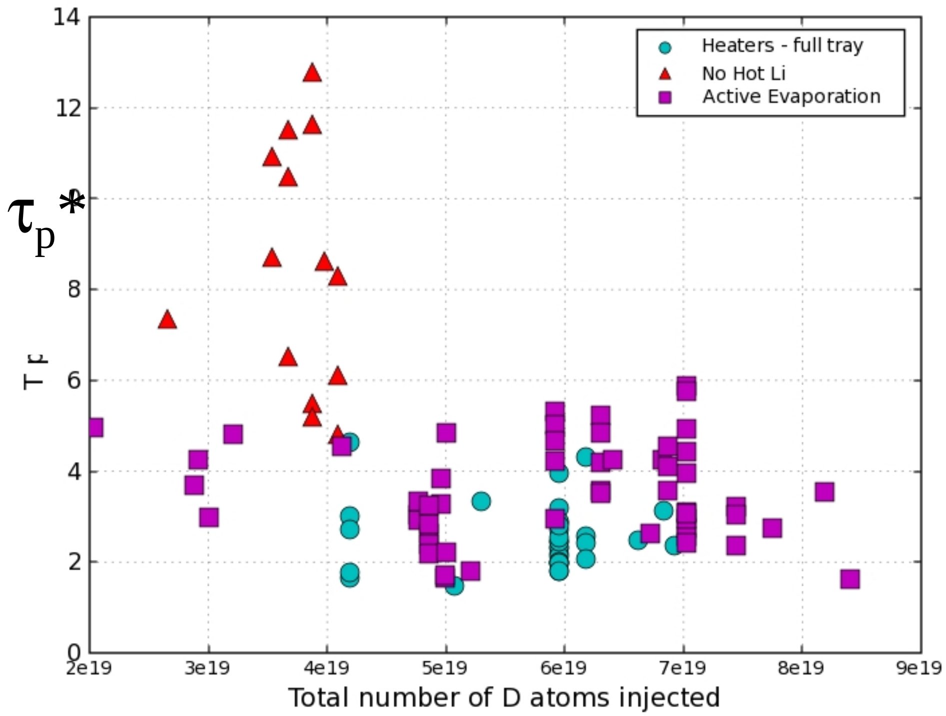
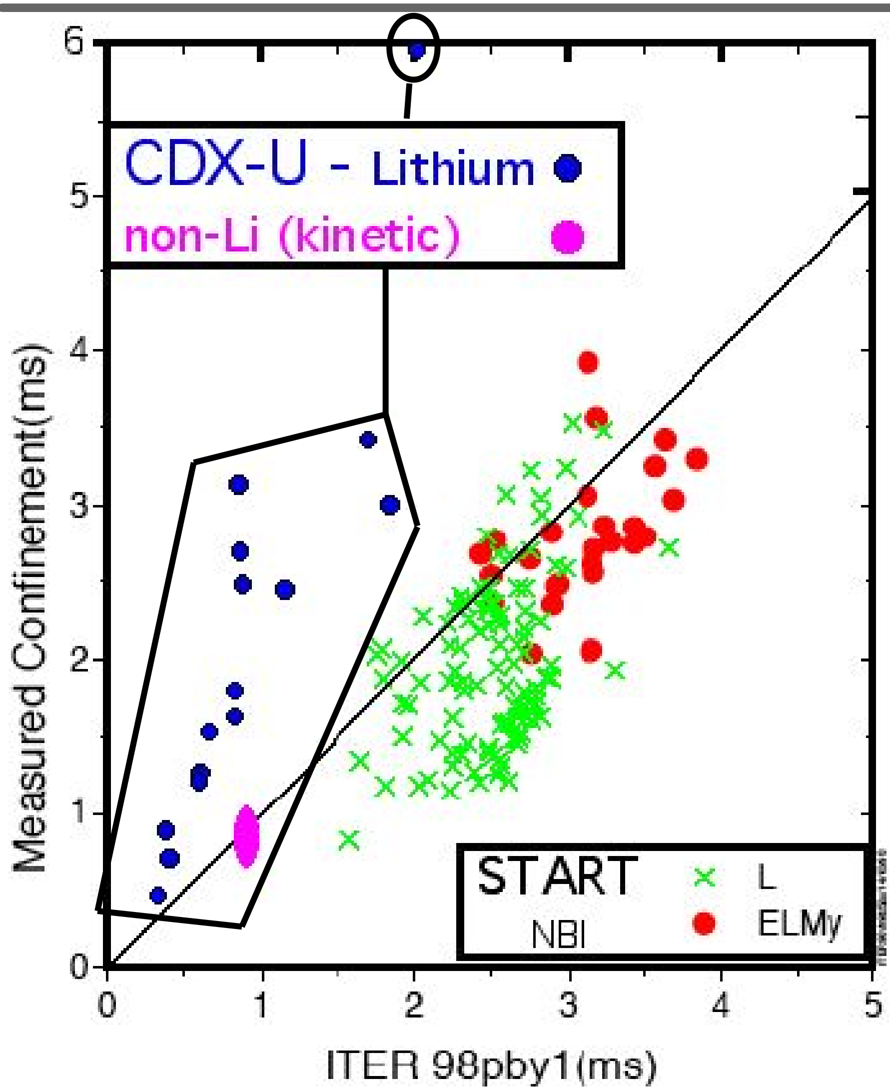
CDX-U - Full wall coatings + liquid lithium produced very high particle pumping rates

- The effective particle confinement time in the presence of recycling, $\tau_p^* \equiv \tau_p/(1\text{-}R)$, is reduced dramatically with liquid lithium limiters and wall coatings
  - τ_p^* too long to measure in the complete absence of lithium wall coatings



- Particle pumping rate in CDX-U is 1 - 2 $\times$ 10^{21} part/sec.
- Sufficient to pump a TFTR supershot
  - But the active wall area in CDX-U is only 0.4 m^2
  - *~Two orders of magnitude* less than the active wall area in TFTR during lithium wall conditioning.
- Liquid lithium also eliminated all traces of water
  - Oxygen vastly reduced
- Carbon, other impurities also reduced

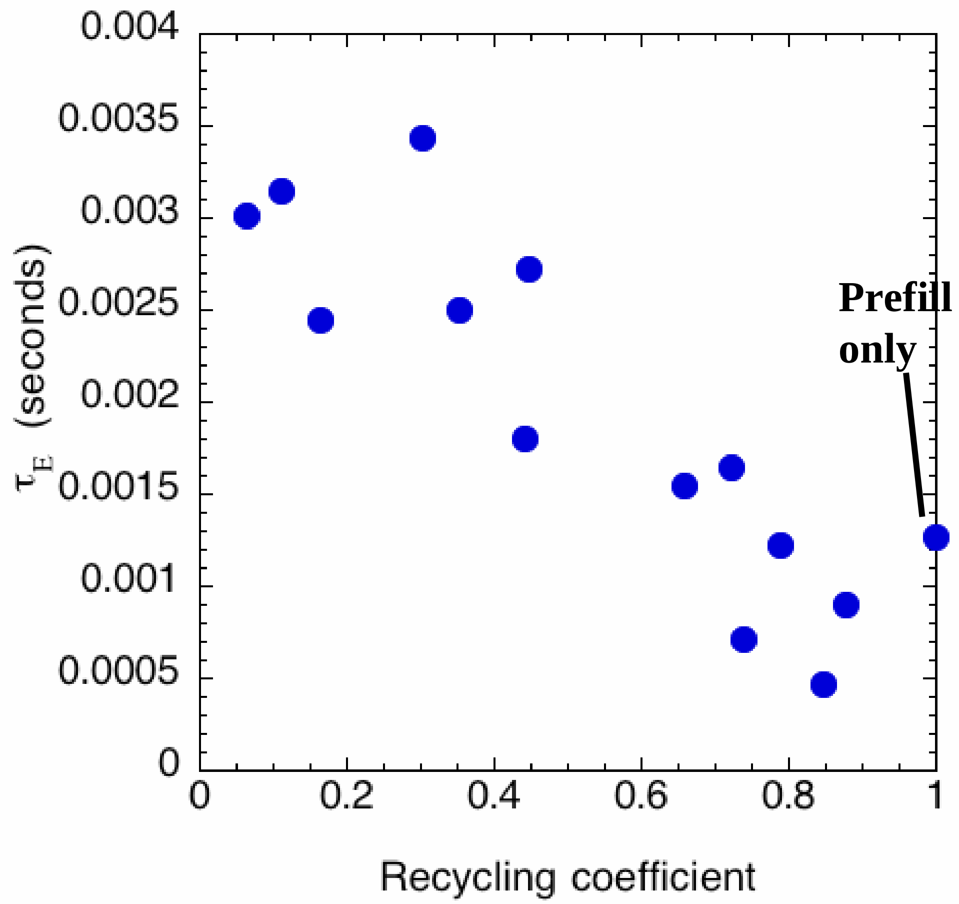
Ohmic confinement is markedly increased



START data from Walsh, APS-DPP98

- Magnetics have been extensively revamped
- Equilibrium and Stability Code (ESC) rewritten to include effects of wall eddy currents (Zakharov)
- Diamagnetic loop, reconstructions used for τ_E
  - *Nonstationary* equilibrium
  - Poynting flux calculation
- Observed confinement times significantly exceed ITER98P scaling
  - ITER98P provides best fit to START data
  - START was similar in size to CDX-U
- New confinement regime for tokamaks

Increased confinement time is correlated with reduced recycling coefficient (R)



- No direct measure of τ_p available
- Here we assume that $\tau_p \sim \tau_E$
  - If $\tau_p > \tau_E$, then we are *overestimating* the recycling coefficient
  - τ_p is unlikely to be much less than τ_E
- Lowest global recycling coefficients ever recorded for a magnetically confined plasma
  - *First magnetically confined plasma* not dominated by wall fueling source

⇒Range in line averaged density: 3.5 - 4.5 $\times$ 10^{18} m^{-3} (prefill only: 6 $\times$ 10^{18} m^{-3})

⇒Range in I_p: 68 - 78 kA (prefill only: 62 kA)

C-MOD Comparison of un-boronized and boronized high-Z operation

Bruce Lipschultz, MIT

Presented by Dennis Whyte, UW

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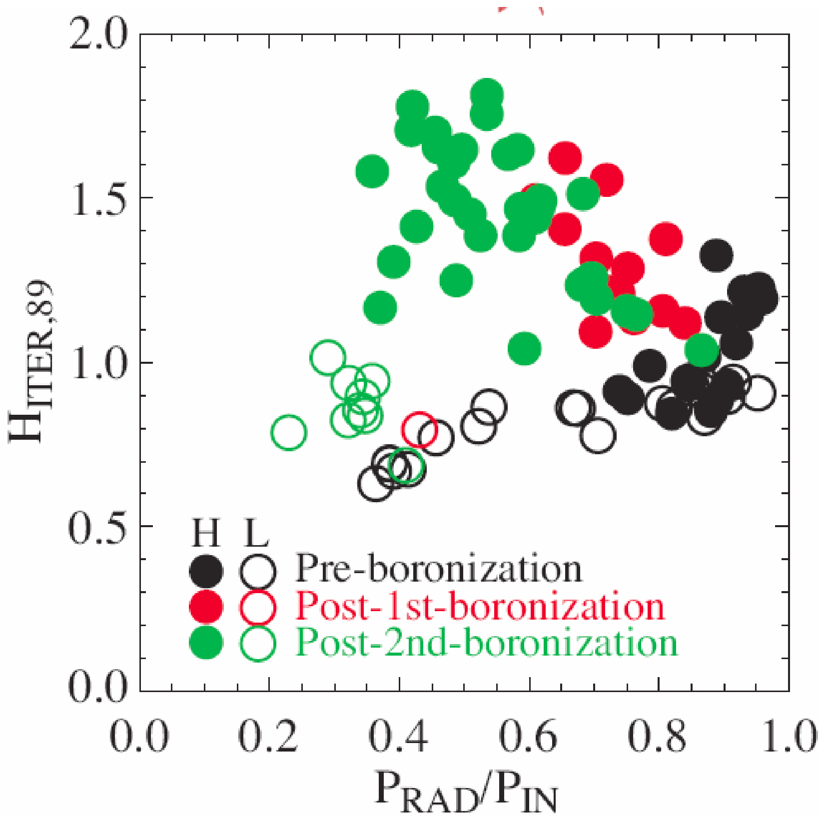
November 2nd, 2005

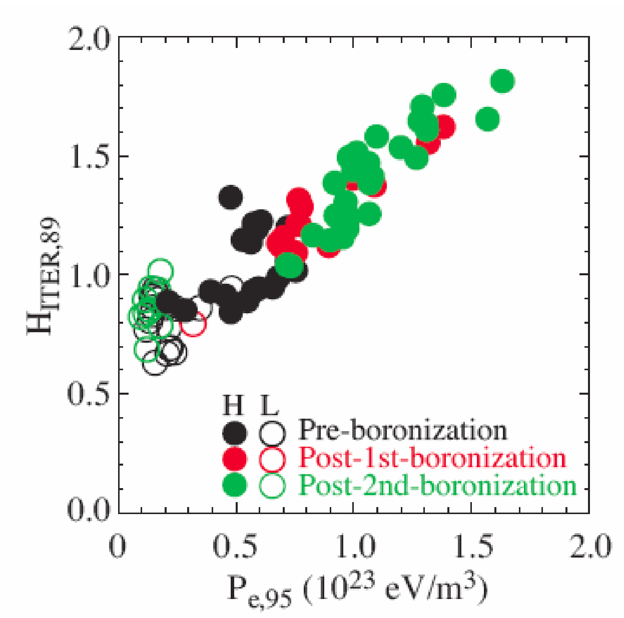
C-Mod campaign dedicated to the comparison of un-boronized and boronized high-Z operation
(B. Lipschultz, et al., APS invited talk)

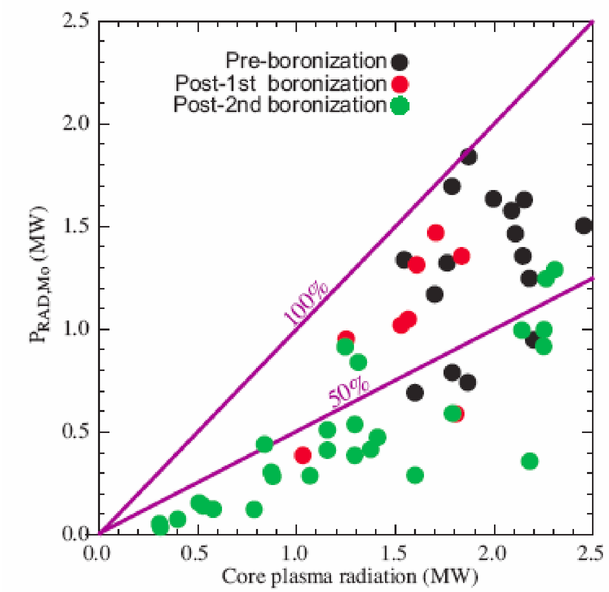
- **Before**: Majority of Mo tiles were covered with thick B layers (~6μm thick)
- **AFTER**: All surfaces cleaned of accumulated boron
  - Surface analysis showed B/(Mo+B) dropping from 99% to 10-20%*
- All BN RF protection tiles replaced with molybdenum.
- Long operational period before boronization to properly characterize un-boronized PFC operation.

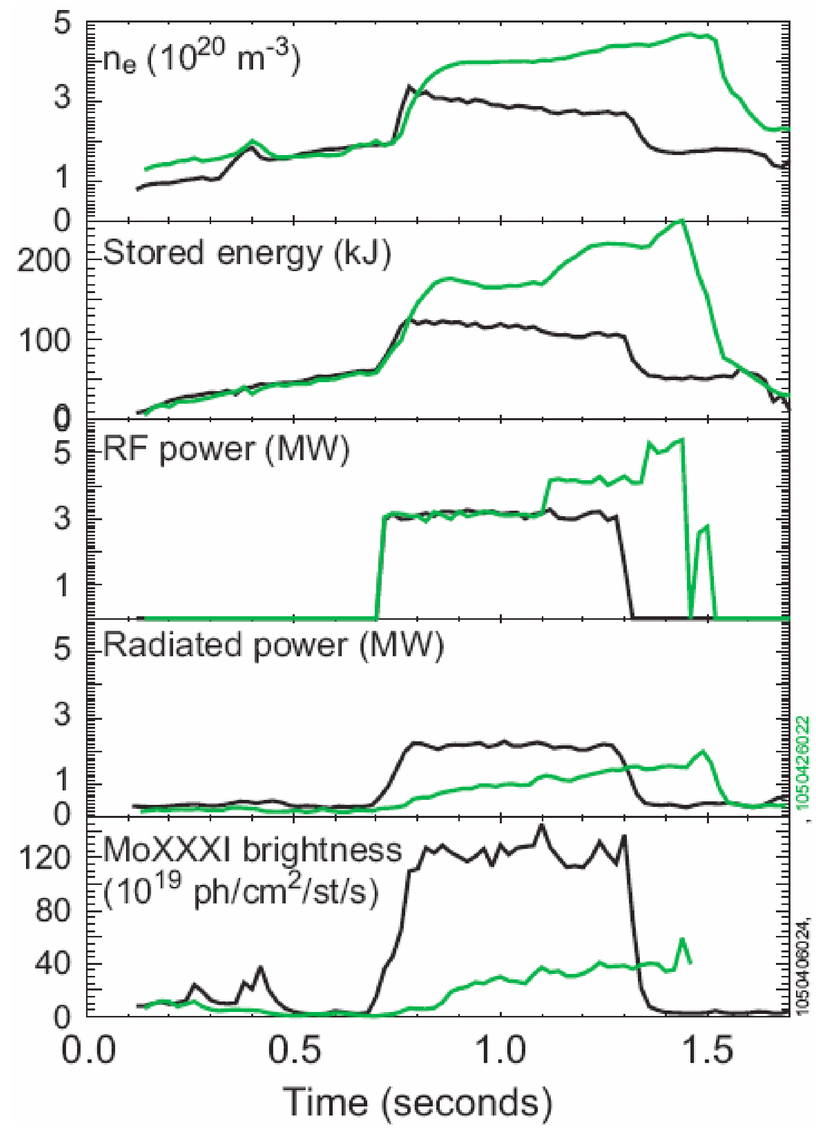


Quality H-mode confinement was not possible with bare Mo wall due to high Mo radiation in core. Boronizations provided good H-mode and world-record tokamak plasma pressure (~2 bar).

H-mode: Core molybendum contamination -> core line radiation losses -> reduced power across pedestal -> reduced confinement with "stiff" core profiles

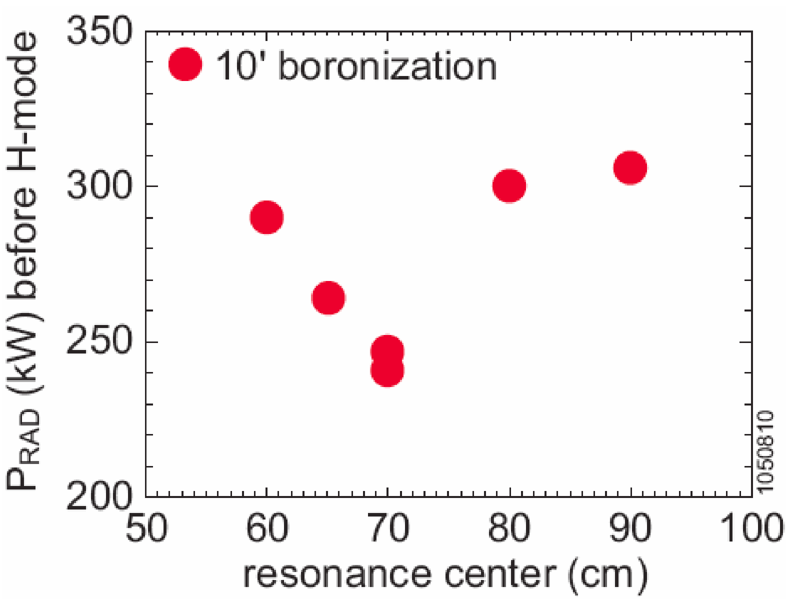


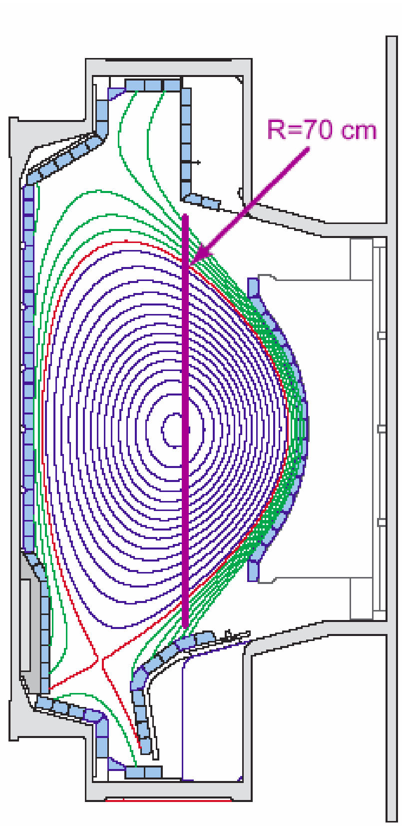


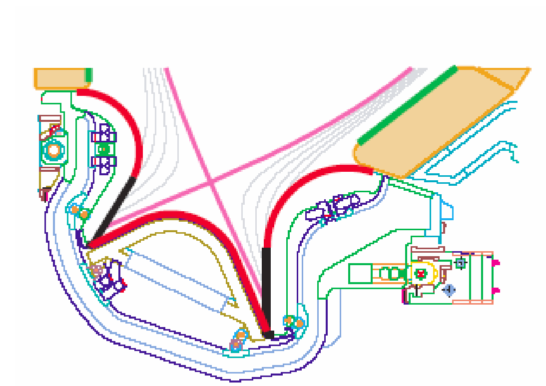


Between-shot boronizations developed on C-Mod: Indicate only specific wall regions responsible for Mo contamination.

Boronization period ~ 1/30 of overnight boronization period



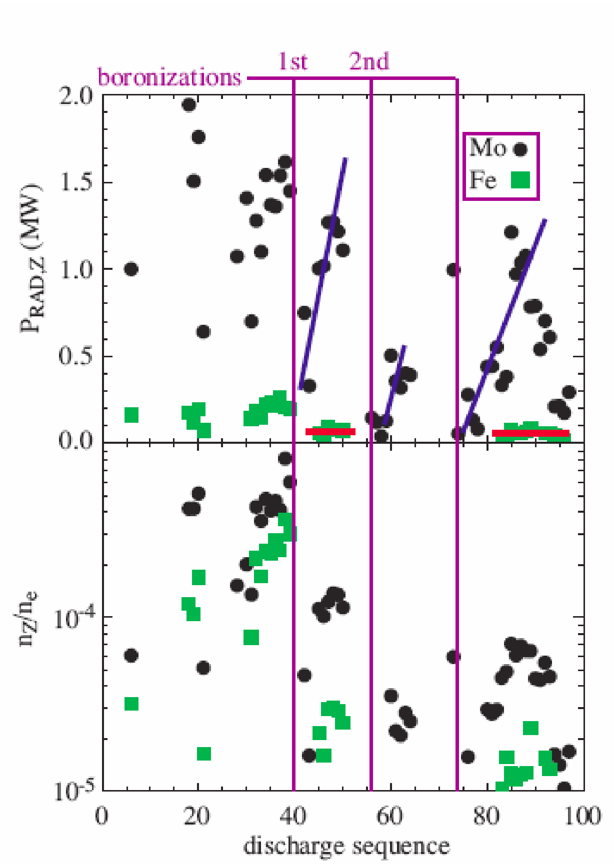


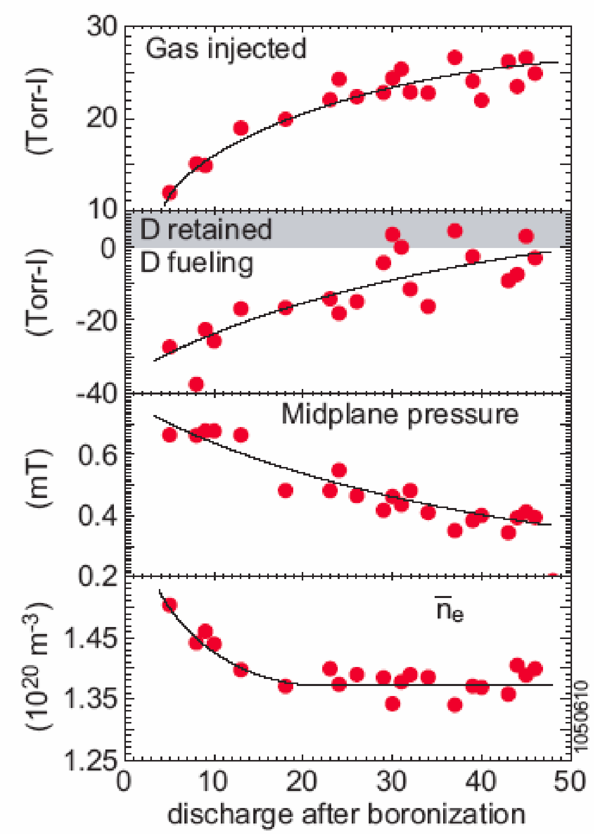


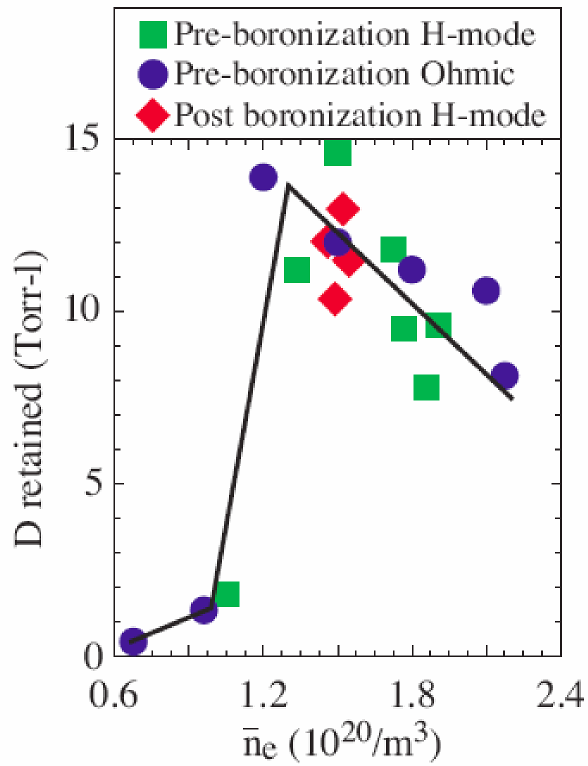
ITER
(W : red)

- **Mo reduction lasts only 1 shot**
- **Surface analysis showed that top of divertor baffle region of B erosion.**

Boronization has transient effects on reducing Mo and recycling: *Surprising result is that nearly pure Mo wall shows ~ 50% D_2 gas retention per shot*





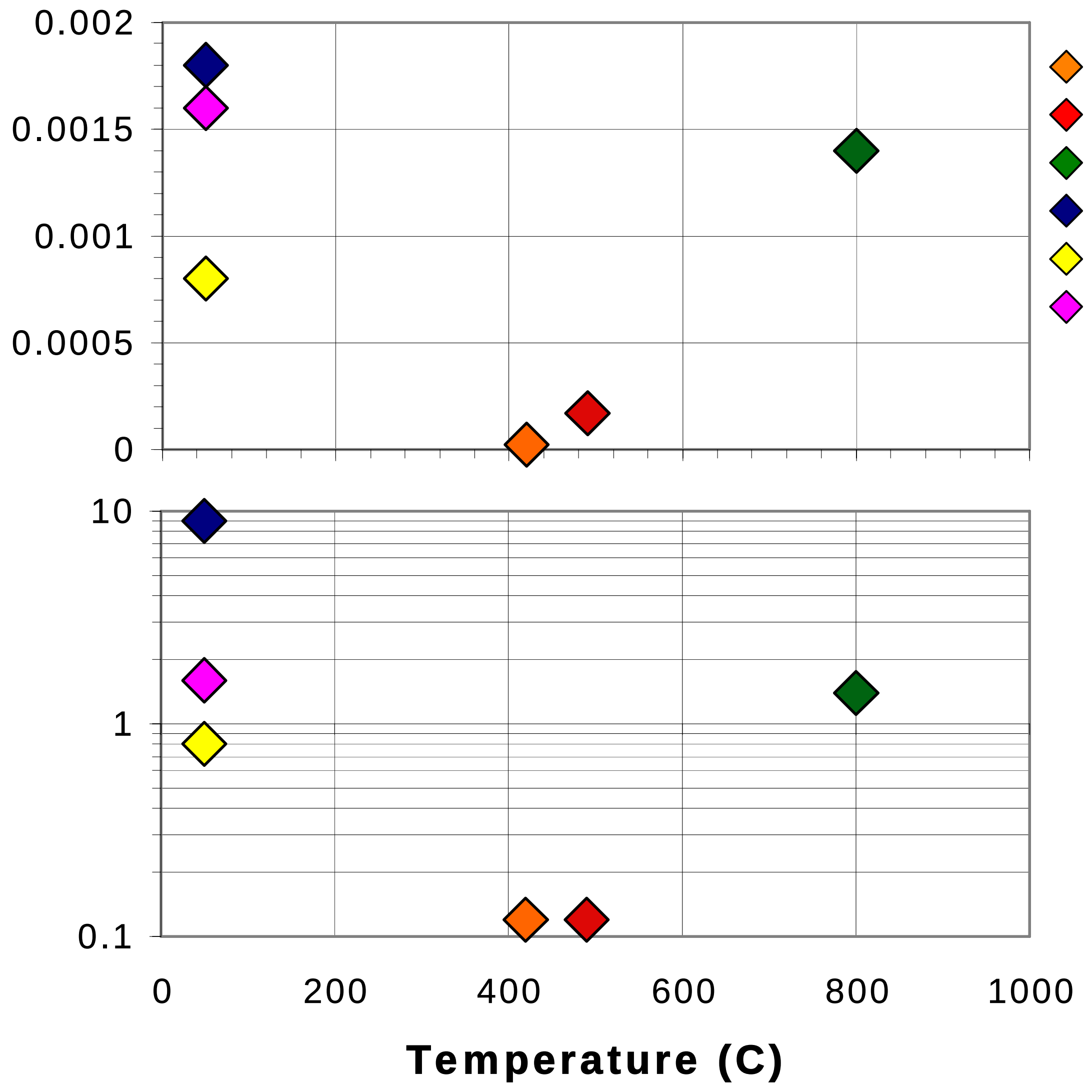


Mo reduction lasts ~30-50 shots

Boronized surfaces first fuel D_2 and then start to Pump after ~50 shots.

About 1/2 of fuelled D_2 is not recovered from wall On a repeatable basis

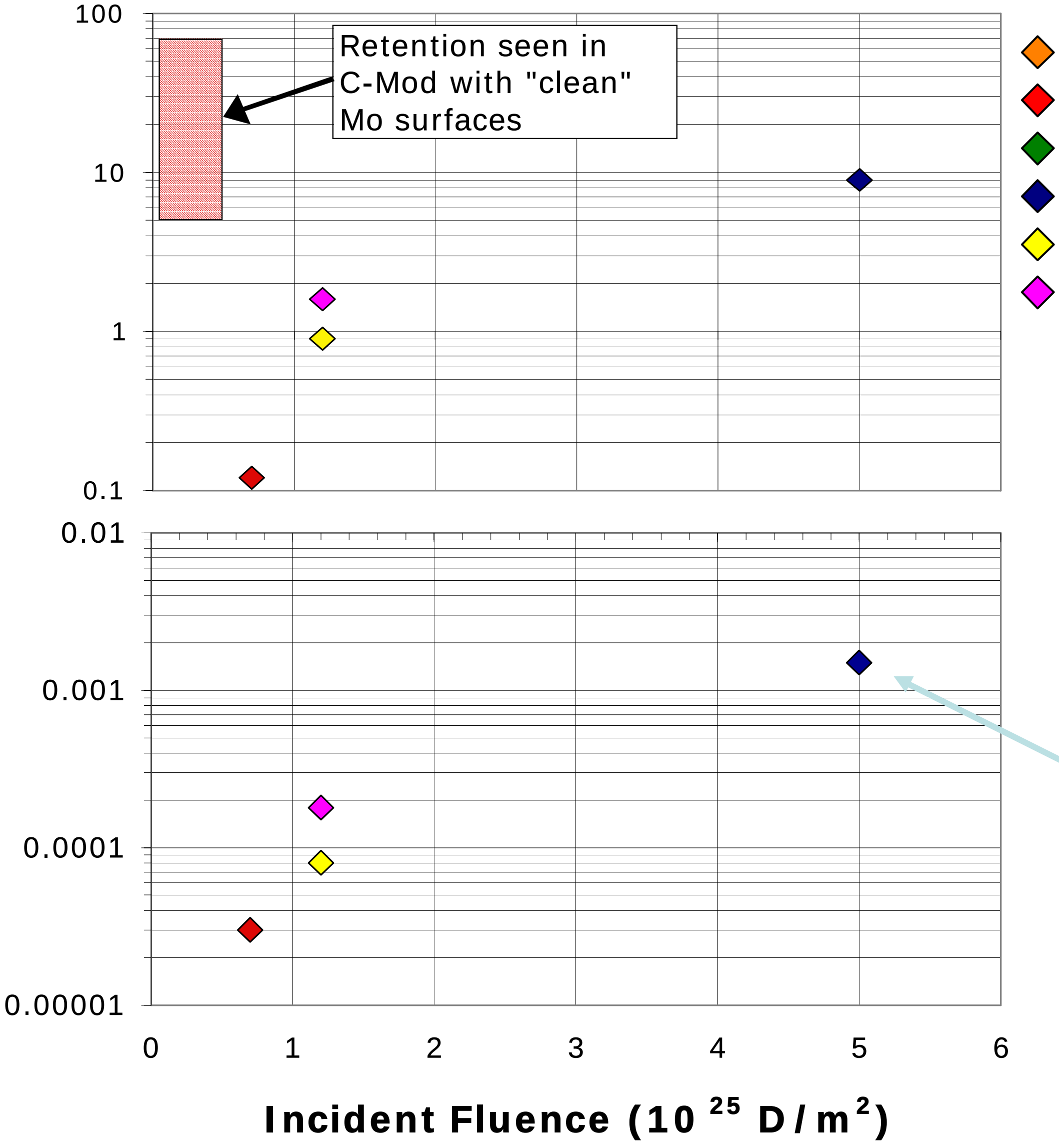
Variety of C-Mod Mo samples exposed to D plasmas in PISCES.
Ion beam surface analysis at U Wisconsin.
Preliminary scoping study of effect of temperature, D fluence, impurities and Mo type on Deuterium retention.



80-90 % of retained D Is found deeper than 0.5 microns up to detection limit ~ 5 microns

Typically, we expect D-retention to decrease with increasing temperature but this is not the case with C1 from surface of outer divertor. The high temperature D-retention seen in C1 may be determined by its higher boron surface content.

Fluence scan shows no sign of saturation for both total retained D fluence and the quantity of "deep" D, but the absolute magnitude of D retention still less than found in C-Mod



Typical fluence for a C-Mod shot is ~10^{23} D/m^2. A fluence of 5×10^{25} D/m^2 (**~500 C-Mod shots**) shows no sign of saturation (sample C1).

Is the trapped deuterium even deeper than the NRA detection limit of ~ 5 microns?

D/Mo > 1000 appm at 5 micron depth

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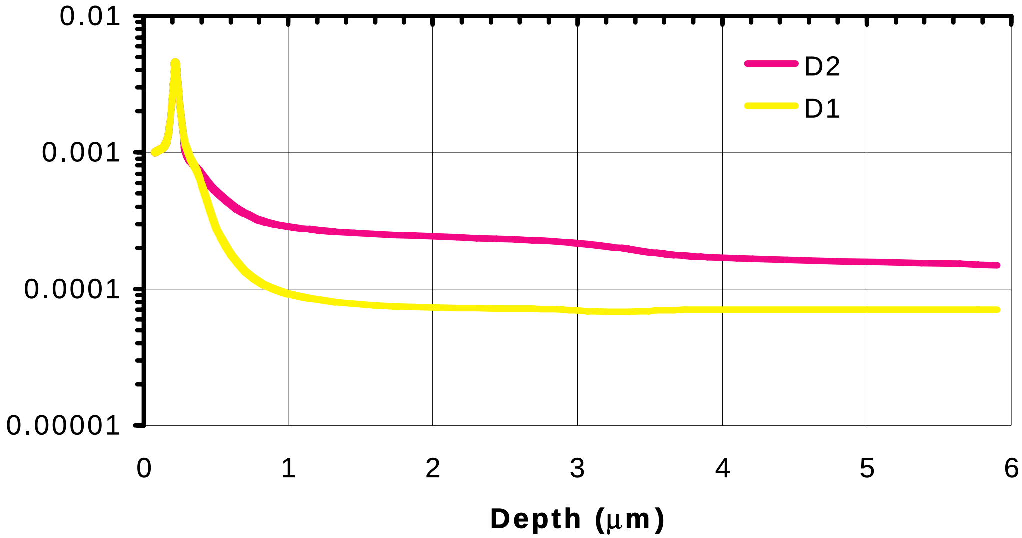
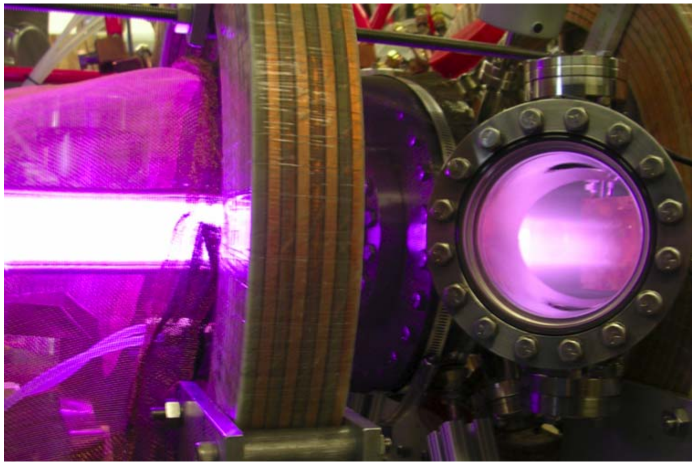
Plasma impurities increase the amount of “deep” deuterium trapped in the Mo samples, suggesting importance of surface damage by heavier ions

Measured D depth profile from ^{3}He-D Nuclear Reaction Analysis



Further experiments to address questions raised about D retention will continue with
Collaboration between PISCES, C-Mod & DIONISOS



Argon Helicon Plasma In DIONISOS

D2: 1 % Neon seeding in exposure plasma (RT)
D1: 0 % “ “
Deuterium measured to depth-limit of NRA

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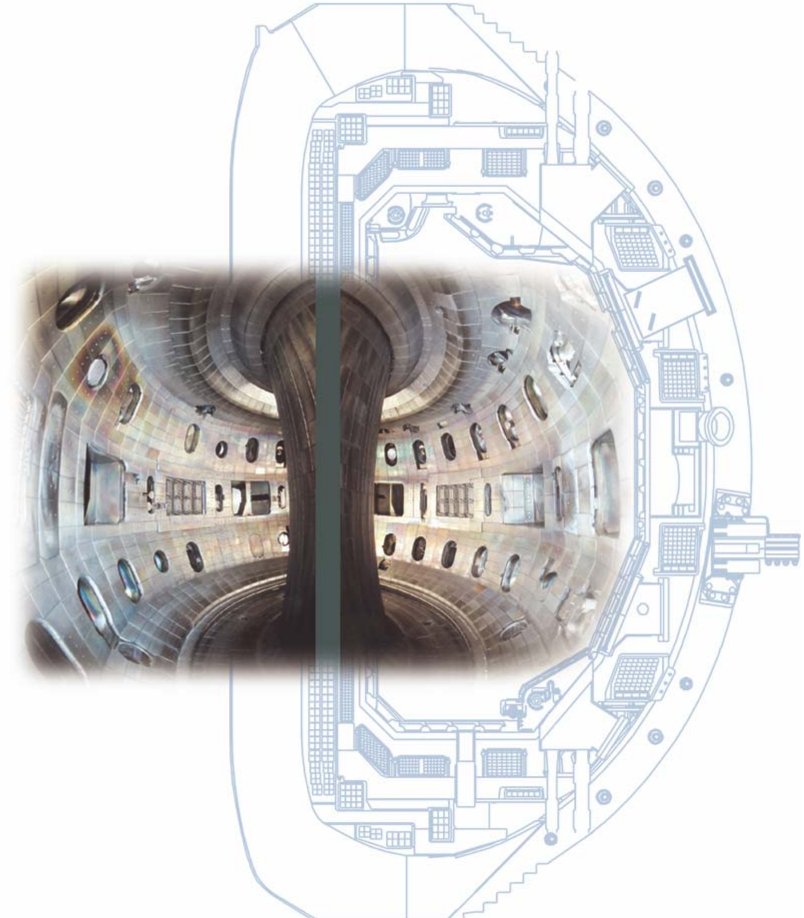
Carbon Co-Deposition Studies in DIII-D L- and H-Mode Plasmas and Implications to the ITER Tritium Inventory

A.G. McLean[a], S.L. Allen[b], W.R. Wampler[d]
P.C. Stangeby[a], W.P. West[e], D.G. Whyte[c],
N.H. Brooks[e], J.W. Davis[a], J.D. Elder[a],
R. Ellis[b], M. Fenstermacher[b], M. Groth[b],
A.A. Haasz[a], K. Holtrope[e], C.J. Lasnier[b],
R.L. Lee[e], G.F. Matthews[h], A. Nagy[i],
V. Phillips[g], G.D. Porter[b], D.L. Rudakov[f],
J.G. Watkins[d], C.P.C. Wong[e]

[a] University of Toronto
[b] Lawrence Livermore National Laboratory
[c] University of Wisconsin
[d] Sandia National Laboratories
[e] General Atomics
[f] University of California, San Diego
[g] FZJ Jülich GmbH/Euratom Institut für Plasmaphysik
[h] Euratom/UKAEA Fusion Association, Culham
[i] Princeton Plasma Physics Laboratory



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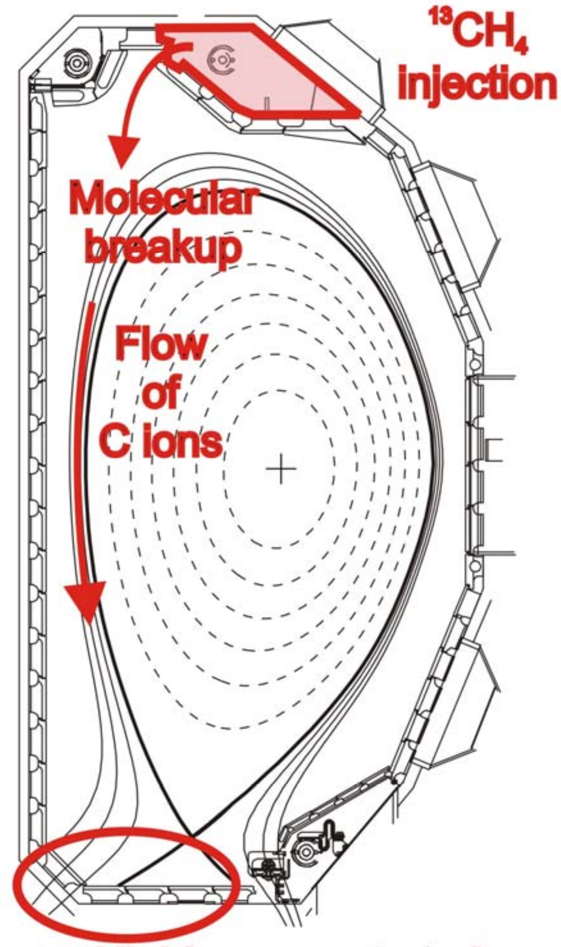
DIII-D
NATIONAL FUSION FACILITY
SAN DIEGO

UNIVERSITY OF TORONTO
Institute for Aerospace Studies

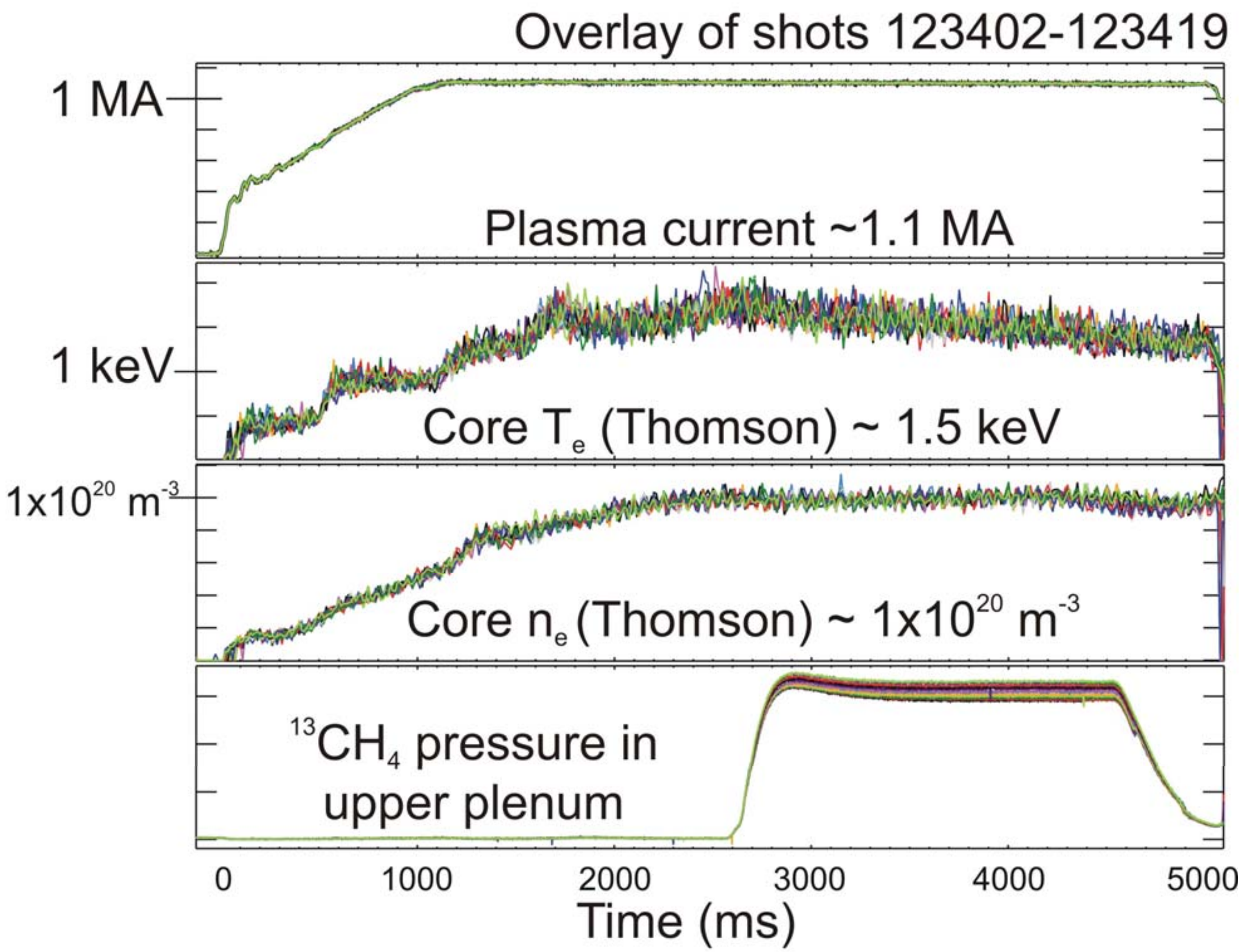
UTIAS

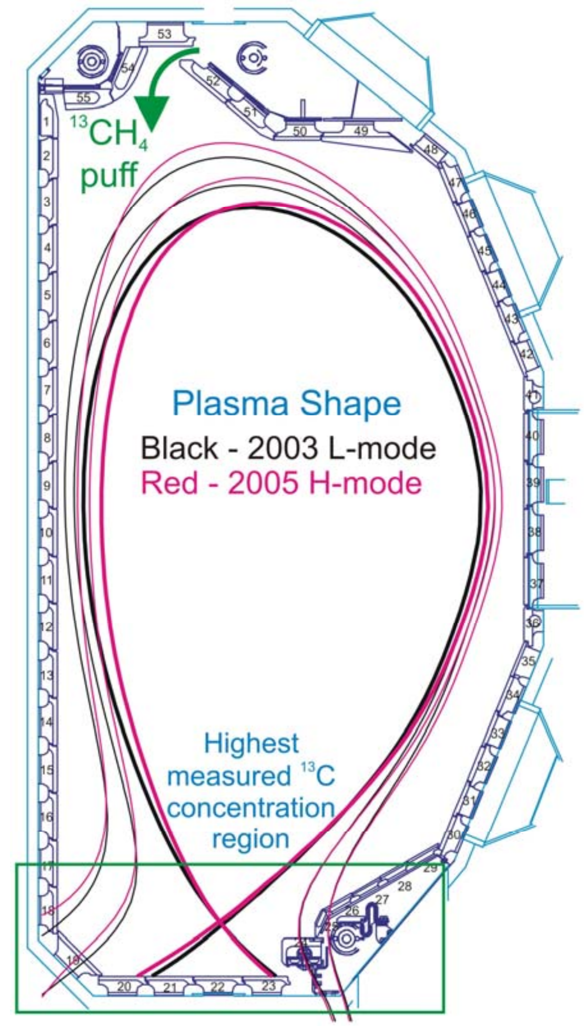
^{13}C tracer injection in DIII-D has been proven to be remarkably revealing

- **2003 experiment in L-mode, 2005 experiment in H-mode with a partially detached divertor (PDD).**
- **Both remarkable similarities and differences in spectroscopic data collected during the experiment, and surface analysis of removed tiles.**
- **In PDD H-mode ^{13}C was found concentrated near the ISP as in L-mode, but also a great deal was found in the private flux zone (PFZ).**
- **Data and simulation so far consistent with model of fast SOL flow towards inner divertor, as found in L-mode.**
- **Tiles removed in 2005 currently being analyzed by IBA at Sandia National Laboratories. Further study by proton-induced γ-ray emission at the University of Wisconsin, oxygen baking at the University of Toronto is planned.**



^{13}C was injected into 17 identical H-mode discharges before DIII-D vent

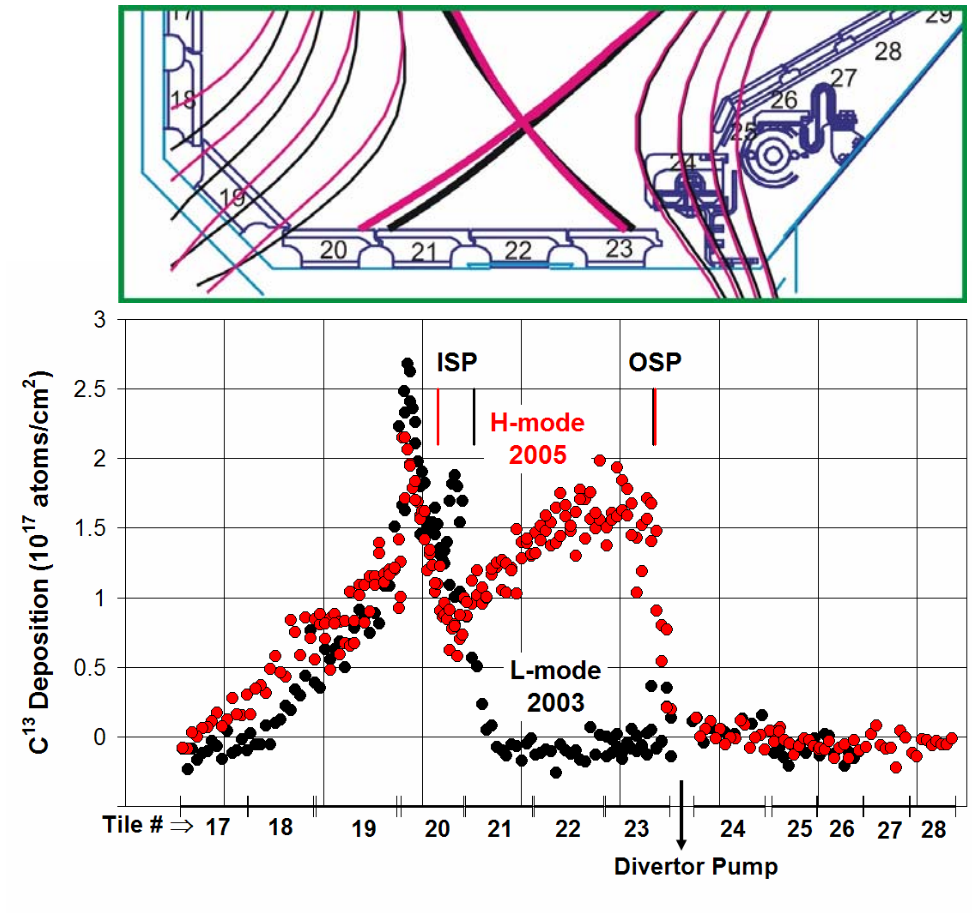




^{13}C deposited inboard of the inner strike point nearly identical in shape to L-mode, however significant deposition found in PFZ

$^{13}C(^{3}He,p)^{15}N$ nuclear reaction analysis
W.R. Wampler, Sandia National Laboratories

- **^{13}C deposition on graphite tiles was measured by IBA**
  - **high near inner divertor for both L&H-mode**
  - **high in PFZ for H-mode only**
  - **low elsewhere**
- **Results at inner strike point consistent with picture that $^{13}CH_4$ is broken up, ionized and swept towards the inner divertor by a fast flow in the SOL.**
- ***However, deposition in the PFZ was not seen in the L-mode experiment: ~equal in total amount to that inboard of ISP***



^{13}C tracer studies on DIII-D confirm scenario of SOL impurity transport, co-deposition, but add new data in a PDD

- **Shift of CIII relative to CII seen in L-mode is direct evidence for fast SOL transport of carbon over the crown of the plasma toward the inside of a divertor tokamak.**
  - **Consistent with highest ^{13}C deposition mainly at the inner divertor**
  - **Deposition on inboard side of ISP in nearly the same in L- and H-mode**

- **In both L- and H-mode, ~30% of injected ^{13}C atoms went to divertor region.**
  - **Measured ^{13}C distribution in L-mode involves neutral deposition at the injection outlet, plus mainly ionic bombardment on the centerpost and inner divertor**

- ***H-mode: Deposition in private flux region requires a transport mechanism not seen in L-mode and not currently well understood.***
  - **Recombination region in partially detached divertor likely plays a significant role (see poster by Elder, this afternoon).**

- **To solve the tritium retention problem caused by co-deposition, these ^{13}C tracer studies are being coupled with *ex-situ* O_2 cleaning studies in Toronto, and possibly *in-situ* on DIII-D.**

- **Proposed future experiments will test deposition with new pumped divertor configuration, and tracer injection with hot (~120-180° C) walls**

DiMES and MiMES in DIII-D

DiMES Recent Results

MiMES Status

D. Rudakov, UCSD
and the DiMES team

PFC e-conference
November 2nd, 2005

Studies of C deposition and D co-deposition in a simulated tile gap - Tile Gap DiMES

Motivation

- Tritium co-deposition/retention is one of the most critical issues for ITER
- Tile gaps are of particular concern
- Altering the tile temperature may affect C deposition and T co-deposition rates
- In DIII-D D co-deposition studied as a proxy for T

Results

- Two exposures were performed under similar plasma conditions, first at ~ 30º C, second at 200º C
- C deposition inside the gap was ~ 2 - 4 times lower in the heated exposure
- **D co-deposition inside the gap was an order of magnitude lower in the heated exposure**
- Net erosion rate of ~ 3 nm/sec was measured on the plasma-facing surface of the heated sample

Modeling: started at ANL (J. Brooks) and Max-Planck-Institute, Germany

Tests of ITER-relevant diagnostic mirrors in a tokamak divertor - Mirror DiMES

Motivation

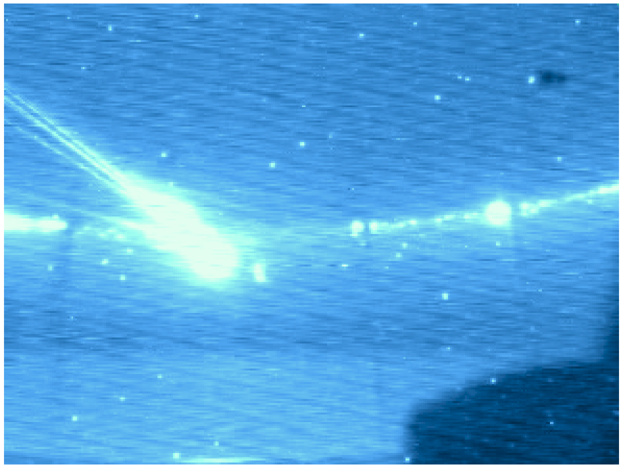
- Optical mirrors are foreseen in ITER for many diagnostics
- Mirrors in the ITER divertor will likely suffer from deposition
- Dedicated experiments in tokamak divertors urgently needed

Results

- We performed first ever tests of ITER-relevant Molybdenum mirrors in a tokamak divertor under well-diagnosed plasma conditions
- Two exposures were performed under similar plasma conditions, first at ~ 30º C, second between 140º C and 80º C
- Non-heated mirrors suffered from carbon deposition at a rate up to 4 nm/sec
- Virtually no deposition on the heated mirrors!

Modeling: not yet started, very desirable

Migration of Micron Size Carbon Dust in the DIII-D Divertor - dust DiMES



Motivation

- **Micron size dust is commonly found in tokamaks**
- **Dust can be a serious problem for ITER for a number of reasons**
- **In particular, dust can cause core contamination and degrade performance**

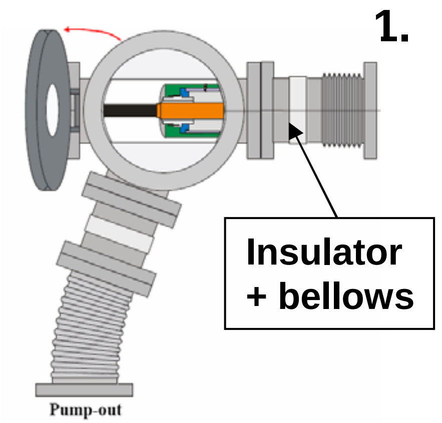
Results

- **We introduced about 30 mg of micron-size dust flakes (5-10 μm in diameter) in the lower divertor of DIII-D using DiMES**
- **In two separate experiments dust was exposed to strike point sweeps of high power LSN discharges in an ELMing H-mode regime**
- **When the outer strike point passed over the dust holder, 1 – 2 % of the total dust carbon content (equivalent to a few million of dust particles) penetrated the plasma core, raising the core carbon density by a factor of 2 – 3**
- **Individual dust particles were observed moving at velocities of 10 – 100 m/s**

Modeling: DustT code (A. Pigarov, UCSD)

Mid-plane Probe Airlock and MiMES

Summary of Motivation and Status

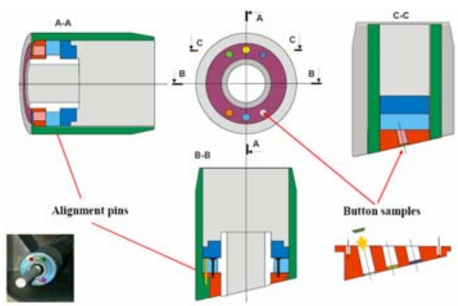


1. Airlock

- ✓ Fast repair of the plunging probe head
- ✓ Using changeable probe heads for specialized physics measurements: flows, Reynolds stress, magnetic fluctuations, ion temperature, etc.
- ✓ Enables MiMES
- ✓ No functionality of the present setup will be lost

2. MiMES (Midplane Material Evaluation Sample)

- ✓ Net erosion/deposition measurements (integrated over exposure time)
- ✓ Tritium retention in the first wall elements (including tile gaps)
- ✓ Complement to the existing DiMES system
- ✓ Bench mark modeling codes
- ✓ Development of new diagnostics



Status: Conceptual Design Review for Airlock/MiMES held on September 19
Final Design Review to be held in November, insulator+ bellows part ordered.